Development of neutron physics models for VVER reactors with NESTLE code

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Introduction

NESTLE code models for VVER-440 and VVER-1000 reactors are developed within the framework of the first part of the Ukraine VVER Special Transient Analysis project [1]. One of the main purposes of this project is to provide support to the In-Depth Safety Analyses that the US Department of Energy is sponsoring in Ukraine. For the development of NESTLE VVER reactor models the compiled information for Rivne NPP unit 1 (VVER-440/B213) and Zaporizhzhya NPP unit 5 (VVER-1000/B320) were used.

Rivne NPP plant data measured during the transient with spontaneous injection of one control assembly, which occurred in January 9, 1998, and Zaporizhzhya NPP plant data measured during the transient with turbine-generator power changes, which occurred in August 20, 1996, were used for NESTLE code verification as source data. Calculations of reactor steady-state conditions before the transients of Rivne NPP unit 1 (17-th fuel load, 213.6 eff. days) and Zaporizhzhya NPP Unit 5 (8-th fuel load, 18.6 eff. days) were performed with Russian codes BIPR-7A and PERMAK-A used in Ukrainian NPPs.

This paper briefly describes the development of the NESTLE VVER reactor models and some results of the NESTLE code verification.

NESTLE VVER reactor models

US neutron kinetics analysis code NESTLE [2] is a FORTRAN77 code that solves the few-group (two or four energy groups) neutron diffusion equation utilizing the nodal expansion method. The NESTLE code models for VVER-440 and VVER-1000 reactors are shown on Fig.1 and Fig.2 respectively.

The NESTLE VVER reactor models are characterized by the following:

- core shape in radial plane is Hexagonal-Z;
- for full scale analysis of specific transients with important local reactivity effects the full core symmetry option is used;
- the number of radial rings of assemblies surrounding central assembly is 12 (two last rings simulate a radial reflector) for NESTLE VVER-440 reactor model and 8 (last ring simulates a radial reflector) for NESTLE VVER-1000 reactor model;
- the number of different radial configurations corresponds to the 10 layers on height of the reactor core and 2 layers of axial reflector;
- for definition of the core material colors the three-dimensional distributions of average burnup calculated by BIPR-7A code are used;
- 4-th group macroscopic cross-section models were developed by WIMSD-5B code [3]; the energy boundaries of the 4-th group representation of neutron spectrum are indicated in Table 1;
- macroscopic models determine the macroscopic cross-sections as a function of color (initial composition of the material within the node), node burnup, rod in or out, coolant density, coolant temperature, effective fuel temperature and soluble poison concentration;
- all assembly discontinuity factors utilized to correct the homogenization errors are derived from results of calculations performed with PERMAK-A code;
- the thermal conditions, predicted by the NESTLE thermal-hydraulic model are used to correct macroscopic cross-sections for temperature and density effects.

Table 1. The energy boundaries of the 4-th group representation of neutron spectrum.

Group number	Energy boundaries	WIMS library groups
1	10 ÷ 0.821 MeV	1 ÷ 5
2	0.821 MeV ÷ 5.53 KeV	6 ÷ 15
3	5.53 KeV ÷ 0.625 eV	16 ÷ 45
4	0.625 ÷ 0.0 eV	46 ÷ 69

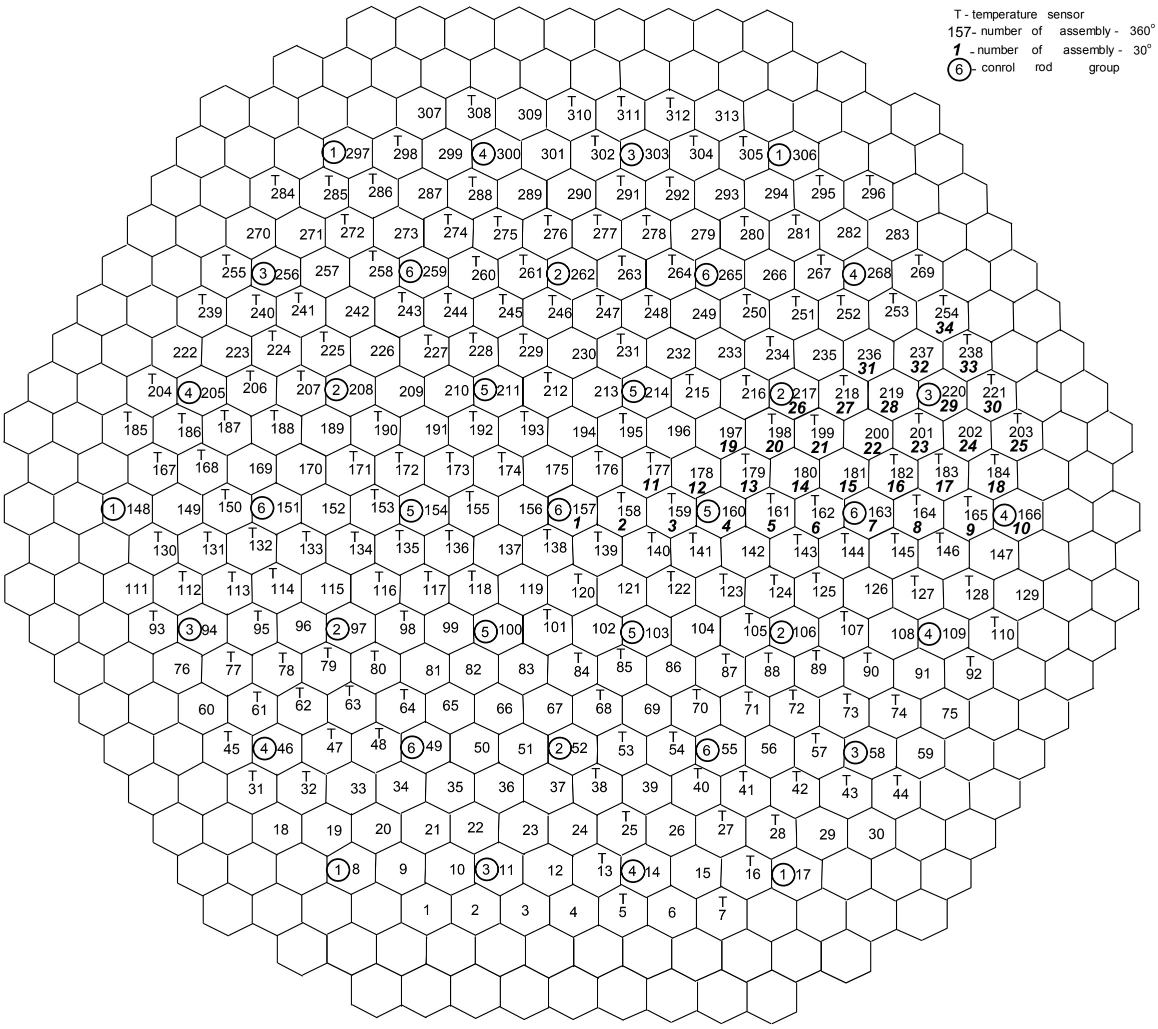

Fig.1. NESTLE code model for VVER-440 reactor.

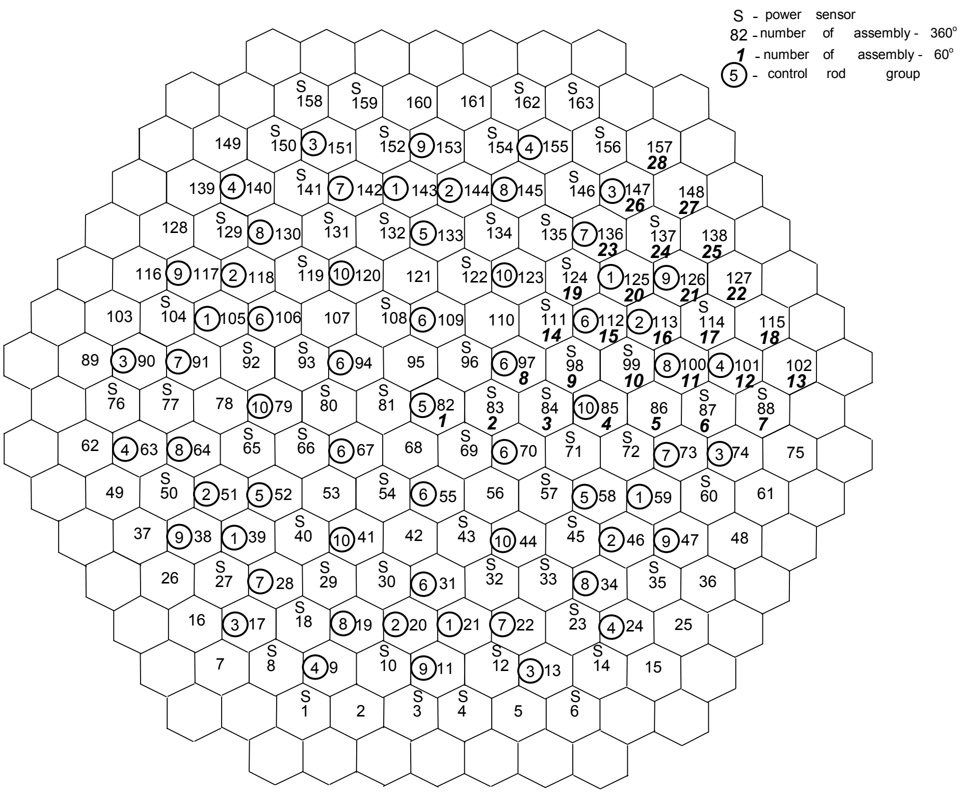

Fig.2. NESTLE code model for VVER-1000 reactor.

Rivne NPP unit 1 transient calculation

An initial event of the Rivne NPP unit 1 transient was the spontaneous injection of one control assembly (assembly №262 of control group №2) from the upper switch up to the second zone (25-50 cm from bottom of reactor core) at the position of working control group 187 cm. After the beginning of falling of control assembly №262 the ARM-5C automatically began to move up on the signal of decrease of neutron power. After 80 sec, at the level of reactor power of about 1332 MW, the operating staff tried to restore the reactor power by ARM-5C in hand-operated mode. After 100 sec, at the level of reactor power of about 1386 MW, the operating staff begins the partial unit power reduction for clearing up of the reasons of initial event. During the transient there was a significant redistribution of power and temperature in the zone of the control assembly injection and in the opposite part of the reactor core.

As source data for verification of the NESTLE code the following plant data measured during the transients were used:

- fuel assemblies' coolant heat-up temperatures;
- main thermalhydraulic parameters: reactor thermal power, reactor mass flow rate, primary coolant temperature of the hot legs and primary coolant heat-up temperature in loops.

The transient parameters were obtained from the transient archives of Rivne NPP unit 1.

Reactor parameters at steady-state conditions before the transients are presented in Table 2. The axial and radial relative power of the reactor core before the transient of Rivne NPP unit 1 obtained with use of BIPR-7A code (P_B) and NESTLE code (P_N) are indicated in Table 3 and Fig.3 respectively. It is visible, that the NESTLE axial and radial relative power are in good agreement with similar BIPR-7A data within the limits of 10 %.

Table 2. Steady-state conditions before the Rivne NPP unit 1 transient.

Parameter	Measurement units	Value
17-th fuel cycle operation time	eff. days	213.6
Reactor thermal power	MW	1359
Active electric power of Turbine-Generators	MW	210.0/210.3
Primary pressure	MPa	12.14
Pressure of Main Steam Headers	MPa	4.25/4.28
Primary coolant temperature in hot legs	^{o}C	295.4
Coolant heat-up temperature	^{o}C	31.0

Table 3. The axial relative power of the reactor core before the Rivne NPP unit 1 transient.

	1	2	3	4	5	6	7	8	9	10
P_N	0.797	1.081	1.131	1.123	1.113	1.110	1.101	1.042	0.900	0.581
P_B	0.805	1.101	1.154	1.149	1.133	1.116	1.087	1.016	0.868	0.572

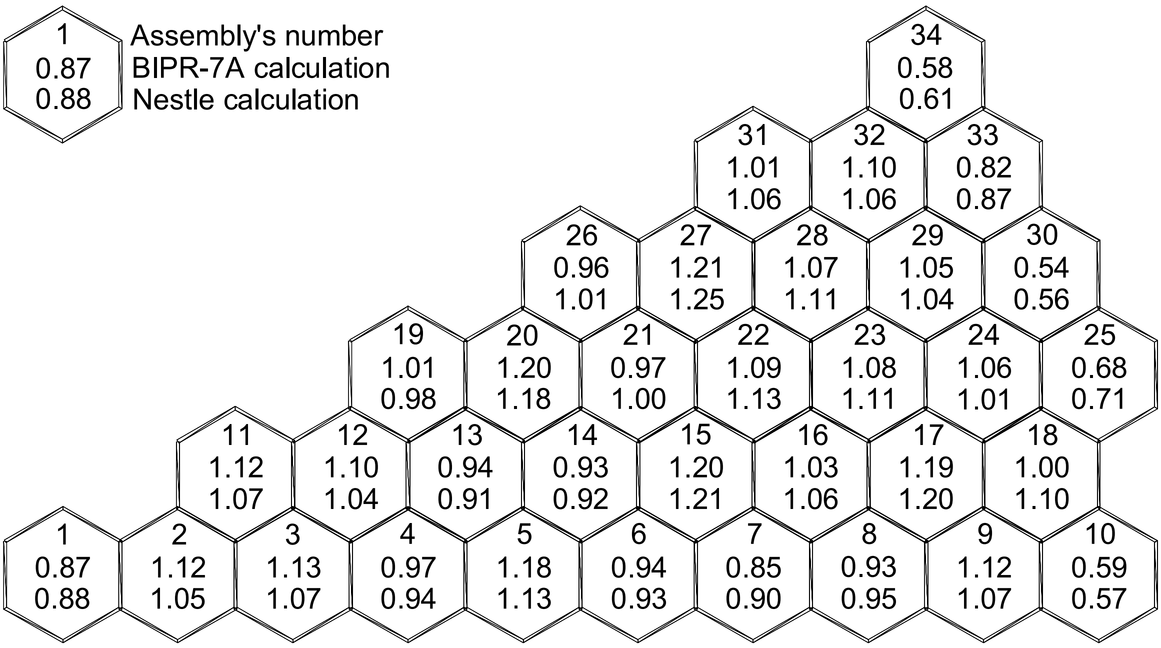

Fig.3. Cartogram of the radial relative power of the reactor core before the transient of Rivne NPP unit 1.

The transient coolant heat-up temperatures in fuel assemblies were calculated by NESTLE VVER-440 reactor model. The NESTLE transient coolant heat-up temperatures in fuel assemblies 229, 231, 246, 247, 276, 277, 291, 31, 44, 62, 73, 116, 124, 173 in accordance with increase of the distance from fuel assembly №262 (source of perturbation) are indicated in Fig.4, 5. The plant transient data are also indicated. The NESTLE transient coolant heat-up temperatures in fuel assemblies agree quite well with plant data. From the figures it is visible, that with increasing distance from the source of perturbation the coolant heat-up temperatures in the fuel assemblies decrease on smaller values, and for assemblies in the opposite part of the reactor core an increase of coolant heat-up temperatures is observed.

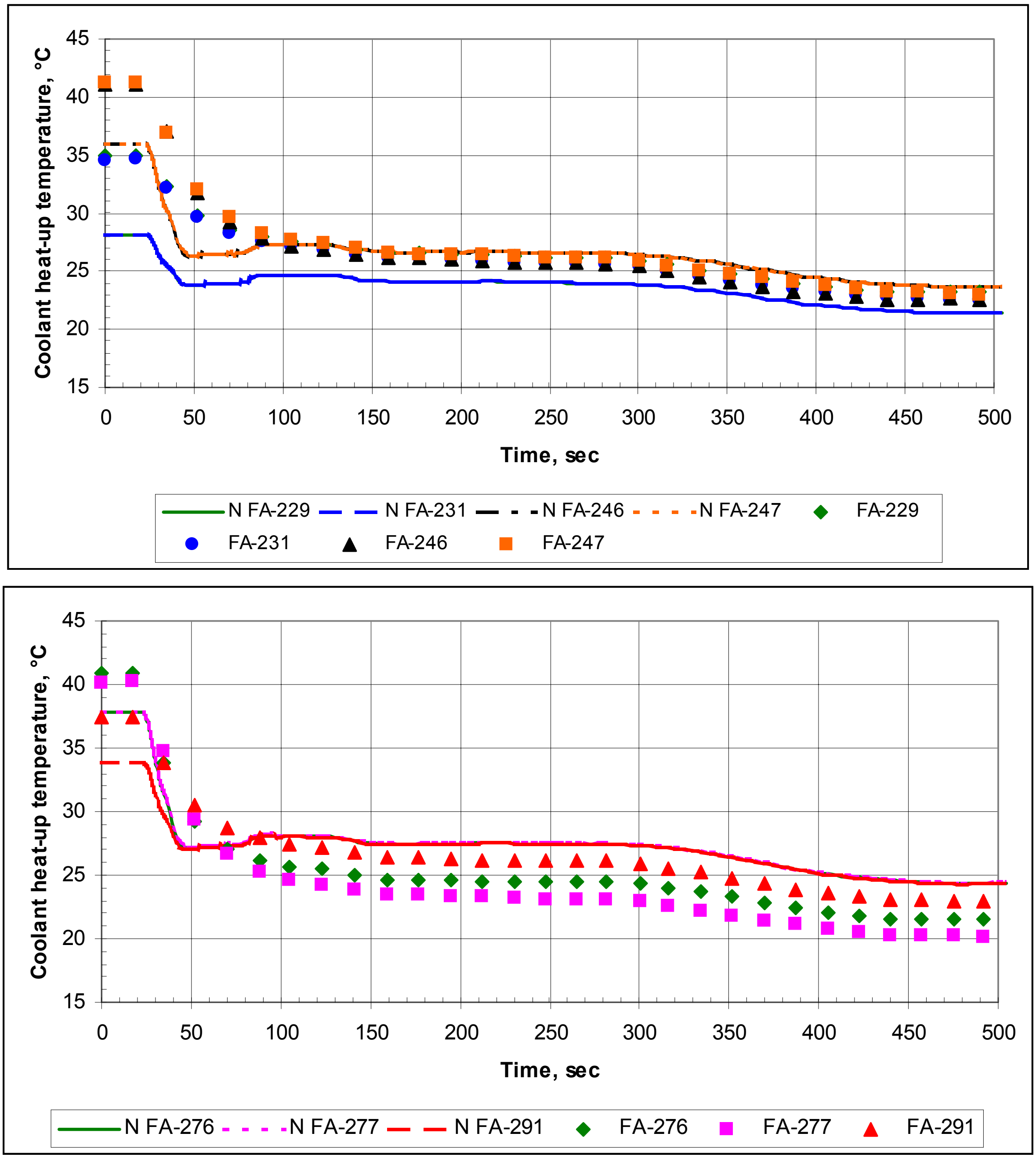

Fig.4. The transient coolant heat-up temperatures
in fuel assemblies 229, 231, 246, 247, 276, 277, 291.

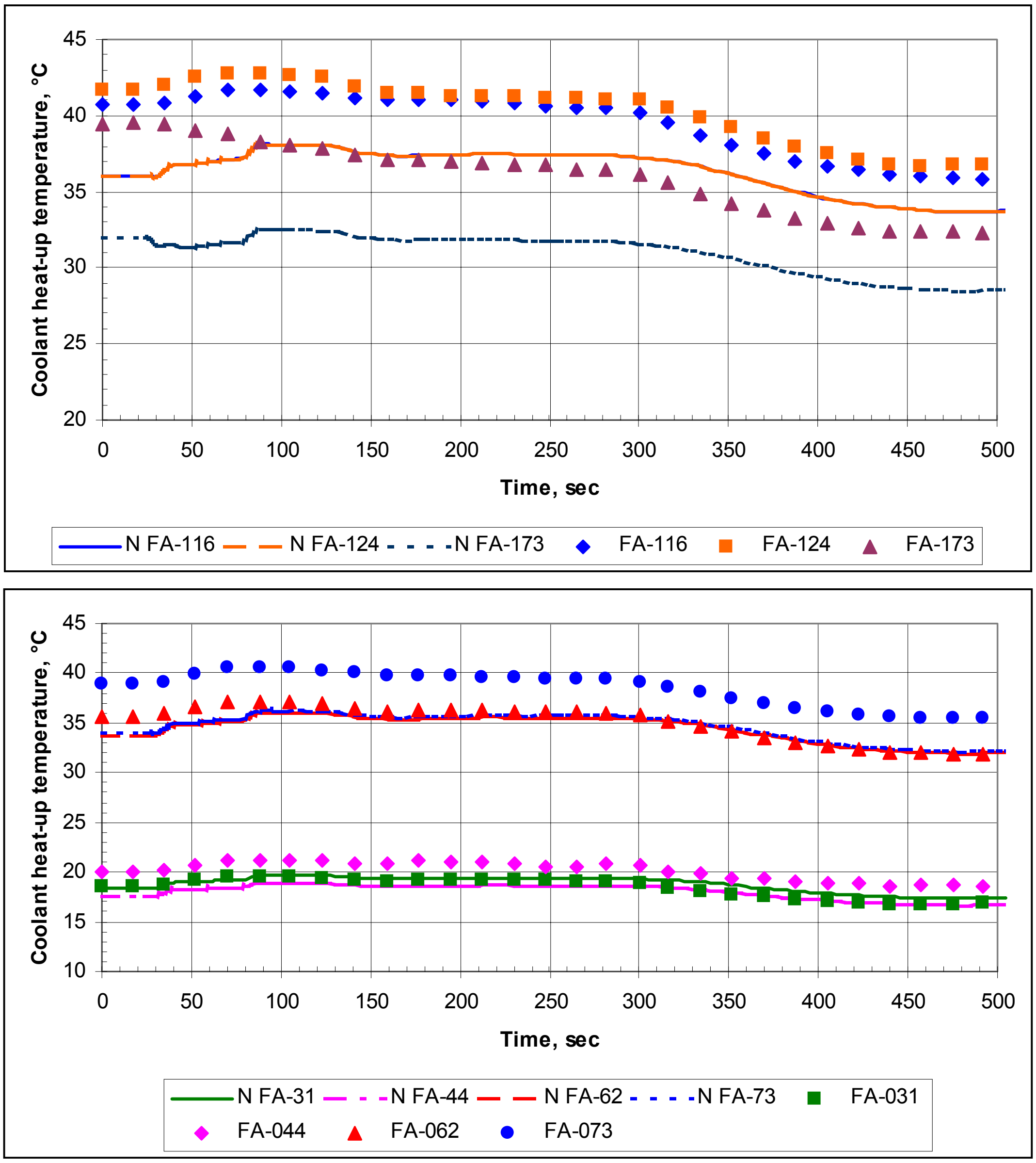

Fig.5. The transient coolant heat-up temperatures in fuel assemblies 31, 44, 62, 73, 116, 124, 173.

Zaporizhzhya NPP unit 5 transient calculation

An initial event of the Zaporizhzhya NPP unit 5 transient was the Turbine-Generator (TG) power decrease from 744 to 649 MW. The most probable reason of the TG power decrease was a high negative signal of electrical frequency control because of a fast outside frequency perturbation. The TG power decrease caused an increase of Main Steam Header pressure and switching of Automatic Reactor Power Controller ARM–5C from a mode “N” to a mode “T”. To support the Main Steam Header pressure, the ARM-5C controller inserted the 10th control rod group in the reactor core. The TG power was restored up to 680 MWt at the end of the transient. During the transient there was a significant redistribution of power and temperature in the reactor core.

As source data for verification of the NESTLE VVER-1000 reactor model the following plant data measured during the transient were used:

- linear powers of the fuel assemblies (at the location where rhodium neutron detector, DPZ, is located);
- main thermalhydraulic parameters: reactor thermal power, reactor mass flow rate, primary coolant temperature of the hot legs and primary coolant heat-up temperature in loops.

The transient parameters were obtained from the transient archives of Zaporizhzhya NPP unit 5.

Reactor parameters at steady-state conditions before the transients are presented in Table 4. The axial and radial relative power of the reactor core before the transient of Zaporizhzhya NPP unit 5 obtained with the use of BIPR-7A code (P_B) and NESTLE code (P_N) are indicated in Table 5 and Fig.6 respectively. It is visible, that the NESTLE axial and radial relative power are in good agreement with similar BIPR-7A data within the limits of 10 %.

Table 4. Steady-state conditions before the Zaporizhzhya NPP unit 5 transient.

Parameter	Measurement units	Value
8-th fuel cycle operation time	eff. days	18.6
Reactor thermal power	MW	2450
Active electric power of Turbine-Generator	MW	744
Primary pressure	MPa	15.64
Pressure of Main Steam Header	MPa	6.09
Primary coolant temperature in hot legs	^oC	314.5
Coolant heat-up temperature	^oC	24.6

Table 5. The axial relative power of the reactor core before the transient of Zaporizhzhya NPP Unit 5.

	1	2	3	4	5	6	7	8	9	10
P_N	0,695	0,983	1,050	1,081	1,103	1,117	1,118	1,094	0,998	0,680
P_B	0,700	0,954	1,042	1,094	1,127	1,140	1,132	1,088	0,973	0,665

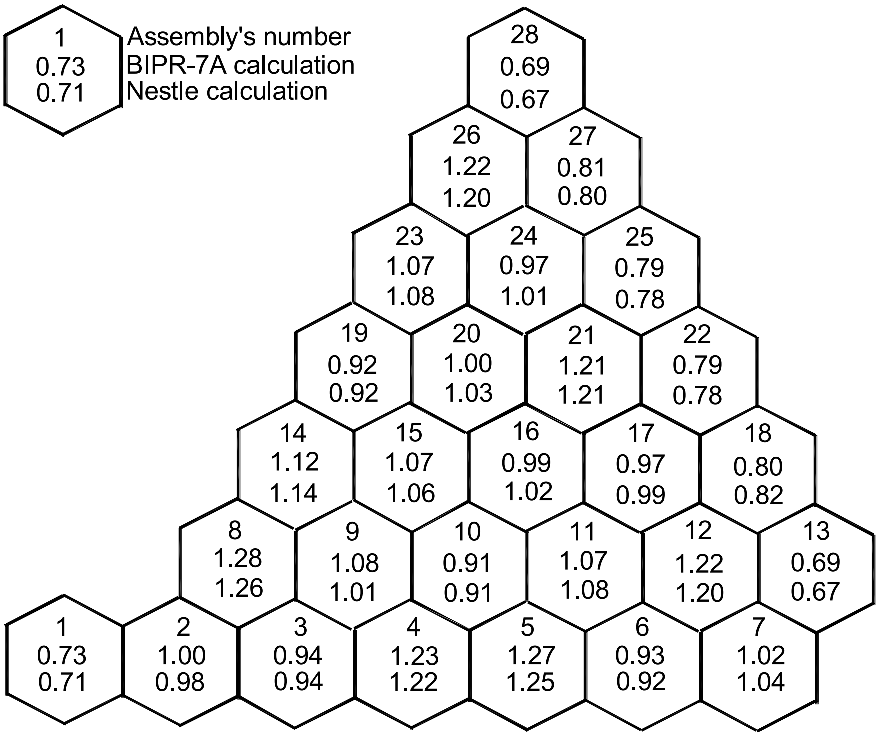

Fig.6. Cartogram of the radial relative power of the reactor core before the transient of Zaporizhzhya NPP Unit 5.

Transient linear powers of the fuel assemblies were calculated with the NESTLE VVER-1000 reactor model. The transient plant data for DPZ-3,4,5 of the fuel assemblies 29, 33, 54, 80, 84, 108, 131, 135 and appropriate NESTLE data are indicated in Fig.7, 8. From Figures it is visible, that the NESTLE transient linear powers of the fuel assemblies agree quite well with the corresponding plant data. The plant transient linear power for symmetrical assemblies where the DPZ is located (in central tube of fuel assembly) specify the magnitude of the scatter of plant data. The NESTLE data for appropriate symmetrical fuel assemblies coincide.

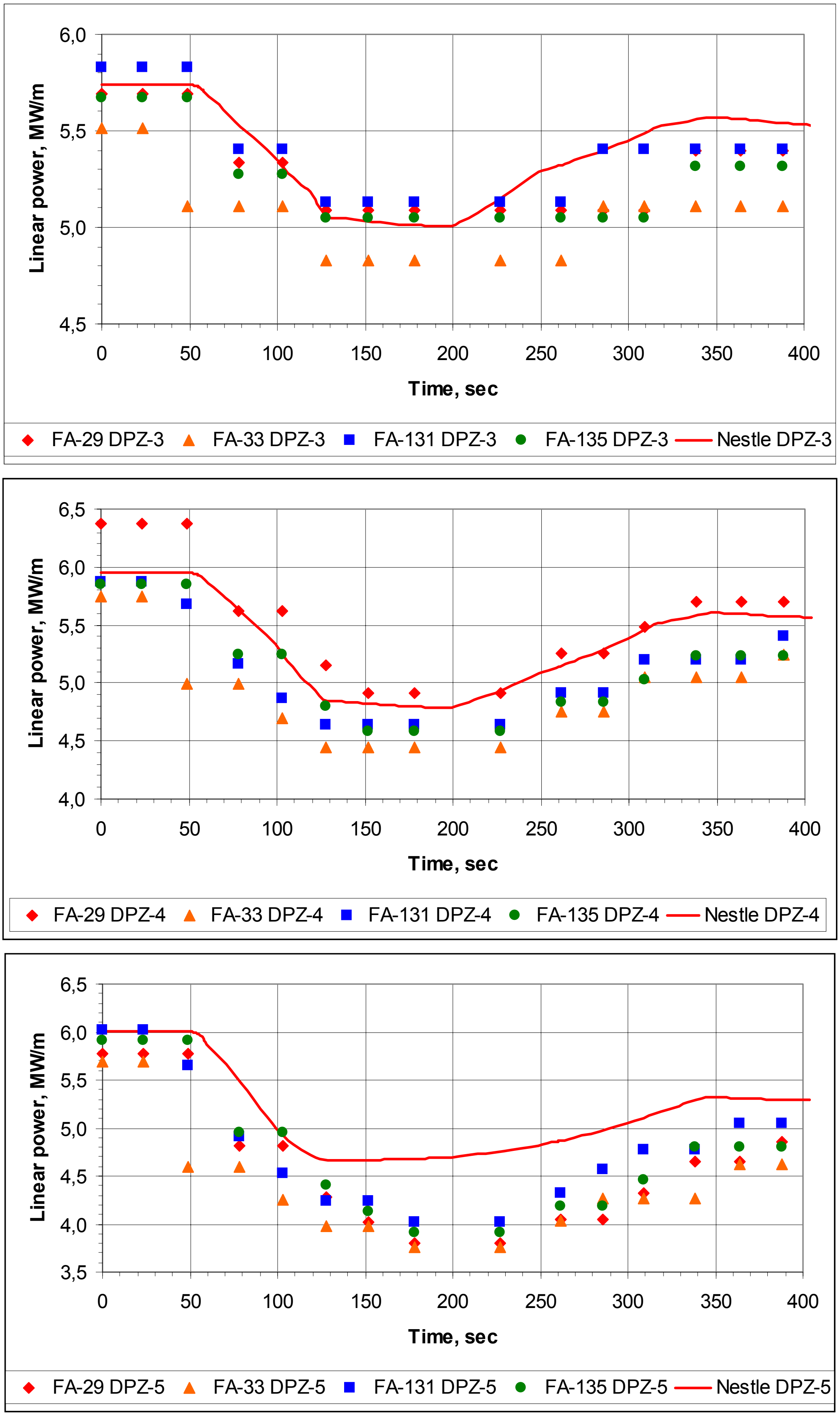

Fig.7. Transient linear powers for DPZ-3,4,5 of the fuel assemblies 29, 33, 131, 135.

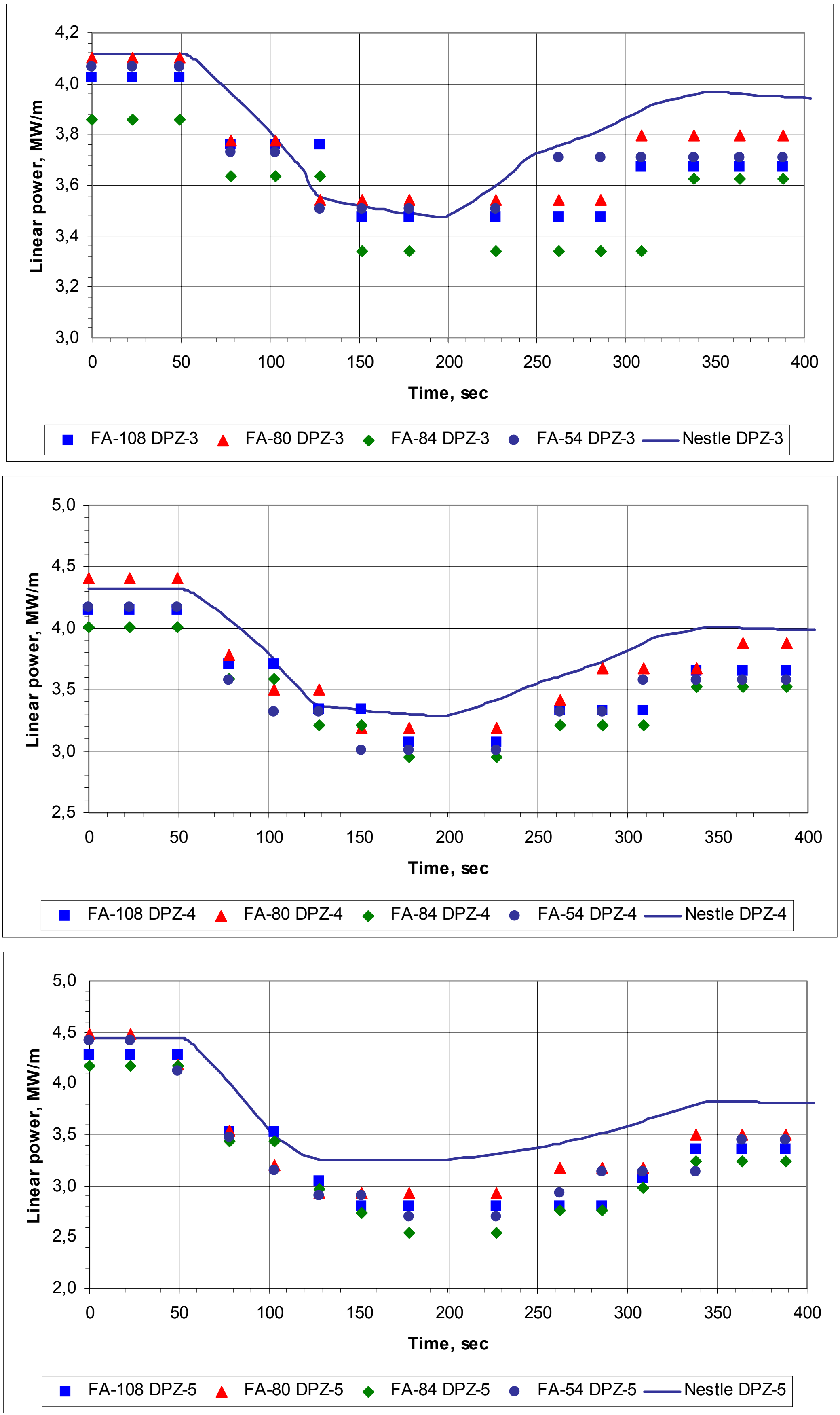

Fig.8. Transient linear powers for DPZ-3,4,5 of the fuel assemblies 108, 80, 84, 54.

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References

1. Kadenko I., Borissenko V., Shkarupa A., Trofimova N., Galchenko V., 1999, "Neutron Kinetics Model Development for VVER-1000 Reactor", Proceedings, 4th International Information Exchange Forum on Safety Analysis for NPPs of VVER and RBMK Types, 11-15 October, 1999; Obninsk.
2. P.J. Turinsky, R.M.K. Al-Chalabi, P. Engrand, H.N. Sarsour, F.X. Faure, W. Guo, NESTLE: A few-group neutron diffusion equation solver utilizing the nodal expansion method for eigenvalue, adjoint, fixed-source steady-state and transient problems. EGG-NRE-11406. June 1994.
3. M.J. Halsall, C.J. Taubman, The WIMS ‘1986’ Nuclear Data Library, AEEW-R 2133, 1986.